**Table I-32.** Summary of intake of waterborne chemical materials of concern based on maximum bounding concentrations listed in Table I-31 compared to Oral Reference Doses.

| Material | Oral Reference Dose | Intake[a] | | |
|---|---|---|---|---|
| | | Proposed Action | Inventory Module 1 | Inventory Module 2 |
| Chromium (VI) | 0.005[b] | 0.00042 | 0.00062 | 0.00065 |
| Molybdenum | 0.005[c] | 0.00026 | 0.00038 | 0.00040 |
| Nickel | 0.02[d] | 0.0010 | 0.0015 | 0.0016 |
| Vanadium | 0.007[e] | 0.0000062 | 0.0000091 | 0.000010 |

a.  Assuming daily intake of 2.0 liters (0.53 gallon) per day by a 70-kilogram (154-pound) individual.
b.  DIRS 148224-EPA 1999, all.
c.  DIRS 148228-EPA 1999, all.
d.  DIRS 148229-EPA 1999, all.
e.  DIRS 103705-EPA 1997, all.

Because the bounding concentration of chromium, molybdenum, nickel, and vanadium in groundwater is calculated to be below the Maximum Contaminant Level Goal or yield intakes well below the respective Oral Reference Doses, there is no further need to refine the calculation to account for physical processes that would limit mobilization of these materials or delay and dilute them during transport in the geosphere.

# I.7  Atmospheric Radioactive Material Impacts

Following closure of the proposed Yucca Mountain Repository, there would be limited potential for releases to the atmosphere because the waste would be isolated far below the ground surface.  Still, the rock is porous and does allow gas to flow, so the analysis must consider possible airborne releases.  The only radionuclide that would have a relatively large inventory and a potential for gas transport is carbon-14.  Iodine-129 can exist in a gas phase, but it is highly soluble and, therefore, would be more likely to dissolve in groundwater rather than migrate as a gas.  Other gas-phase isotopes were eliminated in the screening analysis (Section I.3.3), usually because they have short half-lives and are not decay products of long-lived isotopes.  A separate screening argument for radon-222 is provided in Section I.7.3.  After carbon-14 escaped from the waste package, it could flow through the rock in the form of carbon dioxide.  Atmospheric pathway models were used to estimate human health impacts to the local population in the 80-kilometer (50-mile) region surrounding the repository.

About 2 percent of the carbon-14 in commercial spent nuclear fuel exists as a gas in the space (or *gap*) between the fuel and the cladding around the fuel (DIRS 103446-Oversby 1987, p. 92).  The average carbon-14 inventory in a commercial spent nuclear fuel waste package is approximately 1.37 grams (0.048 ounce) (6.11 curies) (see Table I-5), so the analysis used a gas-phase inventory of 0.122 curie of carbon-14 per commercial spent nuclear fuel waste package to calculate impacts from the atmospheric release pathway.  The waterborne radioactive materials analysis described in Chapter 5, Section 5.4 included the entire inventory of the carbon-14 in the repository in the groundwater release models.  Thus, the groundwater-based impacts would be overestimated slightly (by 2 percent) by this modeling approach.

Carbon is the second-most abundant element (by mass) in the human body, constituting 23 percent of Reference Man (DIRS 101074-ICRP 1975, p. 327).  Ninety-nine percent of the carbon comes from food ingestion (DIRS 148066-Killough and Rohwer 1978, p. 141).  Daily carbon intakes are approximately 300 grams (0.7 pound) and losses include 270 grams (0.6 pound) exhaled, 7 grams (0.02 pound) in feces, and 5 grams (0.01 pound) in urine (DIRS 101074-ICRP 1975, p. 377).

Carbon-14 dosimetry can be performed assuming specific-activity equivalence.  The primary human intake pathway of carbon is food ingestion.  The carbon-14 in food results from photosynthetic processing of atmospheric carbon dioxide, whether the food is the plant itself or an animal that feeds on

the plant. Biotic systems, in general, do not differentiate between carbon isotopes. Therefore, the carbon-14 activity concentration in the atmosphere will be equivalent to the carbon-14 activity concentration in the plant, which in turn will result in an equivalent carbon-14 specific activity in human tissues.

## I.7.1 CARBON-14 RELEASES TO THE ATMOSPHERE

The calculation of regional radiological doses requires estimation of the annual release rate of carbon-14. The analysis based the carbon-14 release rate on the estimated timeline of container failures for the higher-temperature repository operating mode, using the time-dependent mean value of the number of failed waste packages. The expected number of commercial spent nuclear fuel waste package failures in 100-year intervals was used to estimate the carbon-14 release rate after repository closure. The estimated amount of material released from each package as a function of time was reduced to account for radiological decay.

As for the waterborne radioactive material releases described in Chapter 5, Section 5.4, credit was taken for the intact zirconium-alloy cladding (on approximately 99 percent by volume of the commercial spent nuclear fuel at emplacement) delaying the release of gas-phase carbon-14. The remaining 1 percent by volume of the commercial spent nuclear fuel either would have stainless-steel cladding (which degrades much more quickly than zirconium alloy) or would already have failed in the reactor. The cladding failure submodel of the TSPA model also estimates the time of the first perforation through the cladding. Because carbon-14 in gas form as carbon dioxide can migrate through small holes, the time of first perforation was used as the time of release from the carbon-14 from the failed fuel element. A plot of the fraction of the cladding that has been perforated as a function of time after repository closure is shown in Figure I-27.

The amount (in curies) of carbon-14 that would be available for transport, $A_T$, from a waste package at the time it fails is calculated as:

$$A_T = D_F \times F_{FC} \times 0.122 \text{ curies per package}$$

where:

$D_F$ = Time-dependent factor that accounts for radioactive decay (unitless)

$F_{FC}$ = Fraction of perforated cladding (unitless)

The analysis technique calculated the above quantity on a time interval of every 100 years. At each time interval, the amount of carbon-14, $B_T$, available for transport due to further cladding perforations in waste packages that failed previously was also calculated. This amount was calculated as follows:

$$B_T = D_F \times DF_{FC} \times N_{PF} \times 0.122 \text{ curies per package}$$

where:

$DF_{FC}$ = Fraction of cladding that was perforated in the 100-year time interval (unitless)

$N_{PF}$ = Number of waste packages that had failed prior to the current 100-year time interval (unitless)

Rather than conducting a detailed gas-flow model of the mountain, the analysis assumed that the carbon-14 from the failed waste package would be released to the ground surface uniformly over a

100-year interval. Thus, the release rate (curies per year) to the ground surface, $G_S$, for a time interval was calculated as follows:

$$G_S = (N_{CI} \times A_T + B_T) / 100$$

where:

$N_{CI}$ = Number of waste packages that failed in the current 100-year time interval (unitless)

Figure I-28 shows the estimated release rate of carbon-14 from the repository for 80,000 years after repository closure, assuming that the commercial spent nuclear fuel with perforated cladding had released its gas-phase carbon-14 prior to being placed in a waste package. The results in Figure I-28 are based on the Proposed Action inventory. Each symbol in the figure represents the carbon-14 release rate to the ground surface for a period of 100 years. The general downward slope of the symbols is due to radioactive decay (carbon-14 has a half-life of 5,730 years). The symbols indicating near-zero releases (curies per year) indicate that no waste packages failed during some 100-year periods, and the fraction of perforated cladding changed only a small amount. Using this expected-value representation of waste package lifetime, only 1 of 7,860 commercial spent nuclear fuel waste packages would have failed during the first 10,000 years after repository closure. See Section I.2.4 for a description of early waste package failure mechanisms. The second waste package would fail at about 53,000 years after repository closure. By 80,000 years after repository closure, 131 of the 7,860 commercial spent nuclear fuel waste packages would have failed. Using this expected-value representation of the time of first cladding perforation, about 2 percent of the cladding would be perforated in the first 10,000 years. Thus, all releases prior to 50,000 years on Figure I-28 come from a single waste package. The maximum release rate would occur about 1,700 years after repository closure. The estimated maximum release rate would be about 3.3 microcurie per year.

For Inventory Module 1, the number of idealized waste packages containing commercial spent nuclear fuel would increase from 7,860 to 11,754. Using the expected value curves for waste package failure, there would only be 1 waste package failure in the first 10,000 years for Inventory Module 1. Even though the modeled time of the waste package failure is 100 years earlier than for the Proposed Action inventory, the expected value for the fraction of cladding perforated is nearly identical for the two inventory modules during the first 10,000 years. Thus, the maximum release rate to the ground surface is the same and occurs at the same time for both inventory modules. Inventory Module 2 would not add any additional materials expected to contain gas-phase carbon-14, so it would have the same maximum release rate to the ground surface as the Proposed Action inventory.

## I.7.2 ATMOSPHERE CONSEQUENCES TO THE LOCAL POPULATION

DOE used the GENII program (DIRS 100953-Napier et al. 1998, all) to model the atmospheric transport and human uptake of released carbon-14 for the 80-kilometer (50-mile) population radiological dose calculation. Radiological doses to the regional population near Yucca Mountain from carbon-14 releases were estimated using the population distribution described in Appendix G, Section G.2.1, which indicates approximately 76,000 people would live in the region surrounding Yucca Mountain in 2035. The population by distance and sector used in the calculations are listed in Table G-48. The computation also used current (1993 to 1996) annual average meteorology. The joint frequency data are listed in Table I-33.

A population radiological dose factor of $4.6 \times 10^{-9}$ person-rem per microcurie per year of release was calculated using the GENII code. For a 3.3-microcurie-per-year maximum release rate, an 80-kilometer (50-mile) population radiological dose rate would be $1.5 \times 10^{-8}$ person-rem per year. This radiological dose rate represents $7.5 \times 10^{-12}$ latent cancer fatality in the regional population of 76,000 persons each

**Table I-33.** Meteorologic joint frequency data used for Yucca Mountain atmospheric releases (percent of time).[a]

| Average wind speed (m/s)[b] | Atmospheric stability class | Direction (wind toward) | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | S | SSW | SW | WSW | W | WNW | NW | NNW | N | NNE | NE | ENE | E | ESE | SE | SSE |
| 0.9 | A | 0.807 | 0.633 | 0.613 | 0.520 | 0.462 | 0.604 | 0.688 | 0.659 | 0.467 | 0.340 | 0.183 | 0.200 | 0.197 | 0.212 | 0.412 | 0.778 |
| | B | 0.279 | 0.479 | 0.392 | 0.325 | 0.372 | 0.540 | 1.243 | 2.279 | 1.484 | 0.499 | 0.290 | 0.192 | 0.105 | 0.070 | 0.087 | 0.305 |
| | C | 0.113 | 0.105 | 0.064 | 0.017 | 0.015 | 0.020 | 0.041 | 0.157 | 0.122 | 0.067 | 0.055 | 0.020 | 0.012 | 0.020 | 0.009 | 0.032 |
| | D | 0.003 | 0.003 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | E | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | F | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 2.55 | A | 0.099 | 0.073 | 0.026 | 0.020 | 0.026 | 0.017 | 0.023 | 0.061 | 0.041 | 0.029 | 0.023 | 0.017 | 0.029 | 0.029 | 0.052 | 0.096 |
| | B | 0.058 | 0.044 | 0.038 | 0.026 | 0.032 | 0.061 | 0.125 | 0.377 | 0.360 | 0.070 | 0.049 | 0.015 | 0.009 | 0 | 0.009 | 0.017 |
| | C | 0.229 | 0.267 | 0.256 | 0.116 | 0.110 | 0.105 | 0.328 | 1.193 | 2.404 | 0.909 | 0.671 | 0.302 | 0.157 | 0.142 | 0.125 | 0.174 |
| | D | 0.105 | 0.049 | 0.038 | 0.003 | 0.003 | 0.003 | 0.006 | 0.035 | 0.444 | 0.290 | 0.206 | 0.055 | 0.035 | 0.049 | 0.087 | 0.099 |
| | E | 0.003 | 0.006 | 0 | 0.003 | 0 | 0 | 0.003 | 0.003 | 0.003 | 0.006 | 0.003 | 0.003 | 0.003 | 0.003 | 0 | 0.003 |
| | F | 0 | 0.003 | 0 | 0 | 0 | 0 | 0 | 0 | 0.003 | 0.003 | 0 | 0 | 0 | 0 | 0 | 0.003 |
| 4.35 | A | 0.096 | 0.096 | 0.041 | 0.015 | 0.012 | 0.009 | 0.015 | 0.023 | 0.058 | 0.044 | 0.026 | 0.023 | 0.029 | 0.020 | 0.020 | 0.070 |
| | B | 0.052 | 0.087 | 0.041 | 0.023 | 0.006 | 0.026 | 0.078 | 0.261 | 0.305 | 0.131 | 0.076 | 0.017 | 0.006 | 0.003 | 0.009 | 0.032 |
| | C | 0.142 | 0.241 | 0.168 | 0.070 | 0.029 | 0.076 | 0.131 | 0.740 | 1.638 | 0.308 | 0.290 | 0.119 | 0.049 | 0.041 | 0.038 | 0.102 |
| | D | 0.253 | 0.264 | 0.163 | 0.049 | 0.020 | 0.020 | 0.020 | 0.392 | 2.375 | 0.447 | 0.285 | 0.081 | 0.046 | 0.058 | 0.139 | 0.346 |
| | E | 0.006 | 0.017 | 0 | 0 | 0 | 0 | 0 | 0.003 | 0.006 | 0.020 | 0.015 | 0.006 | 0.003 | 0.003 | 0.012 | 0.020 |
| | F | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 6.95 | A | 1.568 | 0.642 | 0.215 | 0.038 | 0.035 | 0.009 | 0.023 | 0.026 | 0.081 | 0.142 | 0.261 | 0.163 | 0.209 | 0.314 | 0.343 | 0.819 |
| | B | 0.682 | 0.552 | 0.067 | 0.003 | 0.006 | 0.006 | 0.023 | 0.058 | 0.348 | 0.325 | 0.267 | 0.131 | 0.078 | 0.093 | 0.078 | 0.256 |
| | C | 0.993 | 0.560 | 0.105 | 0.012 | 0.009 | 0.078 | 0.090 | 0.244 | 0.984 | 0.526 | 0.337 | 0.192 | 0.067 | 0.076 | 0.073 | 0.189 |
| | D | 1.594 | 0.912 | 0.183 | 0.020 | 0.020 | 0.006 | 0.035 | 0.566 | 3.368 | 0.430 | 0.160 | 0.128 | 0.035 | 0.044 | 0.142 | 0.598 |
| | E | 0.735 | 0.366 | 0.067 | 0.012 | 0.006 | 0 | 0 | 0.386 | 2.515 | 0.192 | 0.038 | 0.015 | 0 | 0.015 | 0.064 | 0.804 |
| | F | 0.238 | 0.096 | 0.003 | 0 | 0.003 | 0 | 0 | 0.142 | 1.641 | 0.055 | 0.032 | 0 | 0.003 | 0.003 | 0.029 | 0.796 |
| 9.75 | A | 2.134 | 0.935 | 0.218 | 0.078 | 0.029 | 0.041 | 0.026 | 0.070 | 0.163 | 0.232 | 0.203 | 0.232 | 0.267 | 0.372 | 0.587 | 1.388 |
| | B | 0.865 | 0.627 | 0.081 | 0.009 | 0.003 | 0.017 | 0.020 | 0.046 | 0.319 | 0.267 | 0.154 | 0.131 | 0.070 | 0.052 | 0.113 | 0.302 |
| | C | 0.720 | 0.261 | 0.038 | 0.012 | 0.020 | 0.020 | 0.009 | 0.076 | 0.502 | 0.299 | 0.148 | 0.229 | 0.078 | 0.032 | 0.041 | 0.157 |
| | D | 0.415 | 0.212 | 0.020 | 0.003 | 0.003 | 0.003 | 0.003 | 0.046 | 0.627 | 0.154 | 0.044 | 0.032 | 0.029 | 0.009 | 0.026 | 0.145 |
| | E | 0.029 | 0.006 | 0 | 0 | 0.003 | 0 | 0 | 0 | 0.006 | 0.003 | 0.003 | 0 | 0 | 0.003 | 0 | 0.003 |
| | F | 0 | 0.003 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0.003 | 0 | 0.003 |
| 12.98 | A | 1.661 | 0.706 | 0.418 | 0.322 | 0.247 | 0.244 | 0.366 | 0.343 | 0.407 | 0.380 | 0.302 | 0.299 | 0.357 | 0.537 | 1.083 | 2.038 |
| | B | 0.836 | 0.668 | 0.253 | 0.107 | 0.157 | 0.116 | 0.264 | 0.499 | 0.674 | 0.404 | 0.270 | 0.171 | 0.122 | 0.096 | 0.232 | 0.950 |
| | C | 0.322 | 0.267 | 0.087 | 0.017 | 0.006 | 0.012 | 0.026 | 0.136 | 0.311 | 0.107 | 0.032 | 0.029 | 0.020 | 0.009 | 0.015 | 0.038 |
| | D | 0.006 | 0.006 | 0 | 0 | 0 | 0 | 0 | 0.003 | 0.012 | 0.003 | 0 | 0 | 0 | 0 | 0 | 0.003 |
| | E | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| | F | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |

a.    Source: Adapted from data in DIRS 102877-CRWMS M&O (1999, Appendix B, all).
b.    m/s = meters per second; to convert meters per second to miles per hour, multiply by 2.237.

year at the maximum release rate. This annual population radiological dose rate corresponds to a lifetime radiological population dose of $1.1 \times 10^{-6}$ rem (assuming a 70-year lifetime), which corresponds to $5.3 \times 10^{-10}$ latent cancer fatality during the 70-year period of the maximum release.

The impacts were also calculated for a maximally exposed individual. Given the population data in Appendix G, Table G-48 and the joint frequency data in Table I-33, the maximally exposed individual would reside 24 kilometers (15 miles) south of the repository. An individual radiological dose factor of $5.6 \times 10^{-14}$ rem per microcurie per year of release was calculated using the GENII code for this location. For a 3.3-microcurie-per-year maximum release rate, the individual maximum radiological dose rate would be $1.8 \times 10^{-13}$ rem per year, corresponding to a $9.2 \times 10^{-17}$ probability of a latent cancer fatality. The 70-year lifetime dose would be $1.3 \times 10^{-11}$ rem, representing a $6.4 \times 10^{-15}$ probability of a latent cancer fatality.

### I.7.3  SCREENING ARGUMENT FOR RADON

The uranium placed in the repository would continuously produce radon as a decay product. The longest-lived radon isotope is radon-222, with a half-life of 4 days (DIRS 103178-Lide and Frederikse 1997, p. 4-24). The only potential transport and human exposure pathway for radon would be through the atmosphere because radon would not travel far enough in water to reach an individual before decaying.

A study performed by Y.S. Wu and others (DIRS 103690-Wu, Chen, and Bodvarsson 1995, all) at Lawrence Berkeley National Laboratory calculated gas and heat flow from the mountain due to steam formation and repository induced heating. This study calculated heat and mass fluxes for 57- and 114-kilowatt-per-acre emplacements. The study indicated maximum gas fluxes at the surface of about $2 \times 10^{-7}$ kilogram per second per square meter at the Ghost Dance and Solitario Canyon faults and generally no more that $2 \times 10^{-9}$ kilogram per second per square meter over the remainder of the surface.

The gas flux at the Ghost Dance fault was used to estimate a lower limit for the gas travel time after the waste packages began to fail. The travel times would be longer for a smaller thermal gradient and most waste packages are estimated to remain intact until long after the thermal gradient from the waste emplacement had declined to almost zero. However, this calculation still applies if a waste package failed during the period of highest thermal gradient.

A gas pore velocity, using the estimated gas flux for the Ghost Dance Fault, applicable for gas travel from the repository horizon to the surface, is calculated from the following equation:

$$Vp = Fg / (Da \times Rp)$$

where:

Fg = Gas flux ($2 \times 10^{-7}$ kilogram per second per meter squared)

Da = Density of air (approximately 1.2 kilogram per cubic meter at 20º Celsius) (DIRS 127163-Weast 1972, p. F-11)

Rp = Rock porosity (0.082, unitless) (DIRS 100033-Flint 1998, Table 7, p. 44)

Vp = Pore Velocity (meters per second) = $2.03 \times 10^{-6}$

Travel time from the repository horizon to the surface is calculated from the following equation:

$$T_t = Rd / (Vp \times 86400)$$